Progress Energy

SERIAL: BSEP 07-0024

10 CFR 50.73

FEB 2 3 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Brunswick Steam Electric Plant, Unit No. 2
Docket No. 50-324/License No. DPR-62
Licensee Event Report 2-2006-003

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., submits the enclosed Licensee Event Report. This report fulfills the requirement for a written report within sixty (60) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Randy C. Ivey, Manager - Support Services, at (910) 457-2447.

Sincerely,

BCWaldrep

B. C. Waldrep
Plant General Manager
Brunswick Steam Electric Plant

LJG/ljg

Enclosure:

Licensee Event Report

Progress Energy Carolinas, Inc.
Brunswick Nuclear Plant
PO Box 10429
Southport, NC 28461

IE22

cc (with enclosure):

U. S. Nuclear Regulatory Commission, Region II
ATTN: Dr. William D. Travers, Regional Administrator
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303-8931

U. S. Nuclear Regulatory Commission
ATTN: Mr. Eugene M. DiPaolo, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U. S. Nuclear Regulatory Commission
ATTN: Mr. Stewart N. Bailey (Mail Stop OWFN 8B1) **(Electronic Copy Only)**
11555 Rockville Pike
Rockville, MD 20852-2738

Ms. Jo A. Sanford
Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-051

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	1 OF 6

4. TITLE

Automatic Reactor Scram due to Trip from Neutron Monitoring System

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
12	25	2006	2006	-- 003 --	00	02	23	2007	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
64	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Lee J. Grzeck, Senior Engineer – Licensing	(910) 457-2487

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MO	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On December 25, 2006, at 0539 an automatic reactor scram occurred on Unit 2 due to a Reactor Protection System (RPS) actuation. The unit was operating at 64% reactor power in single recirculation loop operation when the RPS actuated on Neutron Monitoring System Oscillation Power Range Monitors (OPRMs), channels 2 and 4. All control rods properly inserted. Reactor water level reached Low Level 1 (LL1) and Low Level 2 (LL2) as a result of the scram. The LL1 signal caused a Group 2, Group 6, and Group 8 isolation signal. All LL1 actuations occurred as designed. The LL2 signal causes a Reactor Core Isolation Cooling (RCIC) actuation, a High Pressure Coolant Injection (HPCI) actuation, a Group 3 isolation, a Secondary Containment isolation, a Standby Gas Treatment initiation (SBGT), a Control Room Emergency Ventilation (CREV) initiation, a Reactor Recirculation Pump trip, and an Alternate Rod Insertion (ARI) actuation signal. The LL2 condition was reached momentarily, and did not affect all instruments. Further evaluation concluded that the appropriate LL2 isolations and actuations occurred as designed.

This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in manual or automatic actuation of the systems listed in 10 CFR 50.73(a)(2)(iv)(B).

The root cause of this event was determined to be inadequate incorporation of Operating Experience into plant procedures and training.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2006	-- 003 --	00	2 OF 6

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

INTRODUCTION

On December 25, 2006, at 0539 EST an automatic reactor scram occurred on Unit 2 due to a Reactor Protection System (RPS) [JC] actuation. The unit was operating at 64% reactor power in single recirculation loop operation when the RPS actuated on Neutron Monitoring System [IG] Oscillation Power Range Monitors (OPRMs), channels 2 and 4. All control rods properly inserted. Reactor water level reached Low Level 1 (LL1) and Low Level 2 (LL2) as a result of the scram. The LL1 signal caused a Group 2 (i.e., floor and equipment drain isolation valves), Group 6 (i.e., monitoring and sampling isolation valves), and Group 8 (i.e., shutdown cooling isolation valves) isolation signal. All LL1 actuations occurred as designed. The LL2 signal causes a Reactor Core Isolation Cooling (RCIC) [BN] actuation, a High Pressure Coolant Injection (HPCI) [BJ] actuation, a Group 3 isolation (i.e., Reactor Water Cleanup [CE] valves), a Secondary Containment [JM] isolation, a Standby Gas Treatment system (SBGT) initiation [BH], a Control Room Emergency Ventilation (CREV) [VI] initiation, a Reactor Recirculation [AD] Pump trip, and an Alternate Rod Insertion (ARI) actuation signal. The LL2 condition was reached momentarily, and did not affect all instruments. Further evaluation concluded that the appropriate LL2 isolations and actuations occurred as designed. The RCIC system actuation resulted in injection into the Reactor Pressure Vessel (RPV) as designed. The HPCI system actuated, but did not inject into the RPV since reactor water level had recovered.

This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in manual or automatic actuation of the systems listed in 10 CFR 50.73(a)(2)(iv)(B). The NRC was initially notified of this event at 0927 EST on December 25, 2006 (i.e., Event Number 43062).

EVENT DESCRIPTION

Initial Conditions

At the time of the event, Unit 2 was in Mode 1, operating at approximately 64 percent of rated thermal power.

Discussion

On December 11, 2006, monitoring of Reactor Recirculation (RCR) Pump 2A identified that the #2 seal pressure was increasing, indicating signs of degradation and leakage past the #1 seal. This condition raised concerns about the potential failure of both 2A RCR pump seals, and an Operational Decision Making (ODM) plan was developed. The ODM established criteria for shutting down RCR pump 2A and initiated preparations for seal replacement.

On December 22, 2006, RCR pump 2A seal pressure suddenly decreased and a second ODM was completed. Indication of outer #2 seal leakage flow appeared on December 23, 2006, and the decision was

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2006	-- 003 --	00	3 OF 6

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

EVENT DESCRIPTION (continued)

made to secure RCR pump 2A. RCR pump 2A was removed from service at 0225 EST on December 24, 2006, and replacement of the pump seal was planned for December 26, 2006. Unit 2 would be operated in Single Loop Operation (SLO) in the interim.

At 0539 EST on December 25, 2006, while in SLO at 64% reactor power and 52.2% core flow, an automatic scram occurred on Unit 2. The scram was due to a trip signal from the OPRM's channels 2 and 4 Growth Rate Algorithm (GRA). All control rods fully inserted, LL1 actuations occurred, and LL2 was reached on some indicators, causing HPCI and RCIC initiations. RCIC injected, but reactor level did not remain below LL2 long enough for HPCI to inject. SBGT and CREV systems initiated and Reactor Building Ventilation isolated as expected. At the time of the event, the reactor was being operated in an approved area of the Power/Flow map, and within the operating parameters for SLO.

Inward rod movements had been conducted on control rods 26-15 and 26-39 at 2138 EST on December 24, 2006, and at 0350 EST on December 25, 2006, respectively, in accordance with nuclear engineering guidance in order to compensate for xenon burnout and to obtain additional margin to the Maximum Extended Load Line Limit Analysis (MELLLA). A review of data found that small momentary power oscillations had occurred at random intervals during the hour before the scram, however they did not exceed any of the OPRM trip settings. At 0539 EST another momentary oscillation started and led to the GRA trip.

After entering SLO, associated OPRM annunciators alarmed randomly approximately 25 times but did not lock in and cleared within a few seconds. Past experience has been that these alarms periodically occur when at reduced power, so these occurrences were not considered unusual. Numerous Process Computer Event Log alarms were also received with the Average Power Range Monitor (APRM)/OPRM trips and eventually OPRM Cell Growth Rate warnings, but these are not audible, provide indication only, and are not routinely monitored due to the large number of invalid and lower tier data points displayed. No audible Control Room annunciators were received immediately prior to the trip and there were no other indications of any unusual anomalies.

The reactor and associated protection system is designed such that power oscillations are either not possible or can be readily detected and suppressed without exceeding specified fuel design limits. Historically, compliance was achieved by demonstrating that Thermal-Hydraulic Instability (THI) induced neutron flux oscillations were not expected. BSEP initiated several options that provide an appropriate level of protection for stability-related neutron flux oscillations. The THI issue was addressed the following three ways:

- Option III of the OPRM system, which employs a detect and suppress method of instability response, was installed using the Low Power Range Monitors (LPRMs).

EVENT DESCRIPTION (continued)

- The Power/Flow map was revised to identify the OPRM enabled region where the potential for THI is a concern. Additional Power/Flow maps were created for SLO and operation with the OPRMs inoperable.

- A scram avoidance region was added to the Power/Flow maps. This highlights the area (i.e., high power/low flow) of greatest risk for an OPRM trip due to the increased sensitivity of the OPRM settings. Entry into this region requires immediate operator action to exit the region. Since an OPRM trip will cause a reactor scram upon detection of a THI event, this region also is an area of greatest risk of a scram.

As a result of station evaluation and review of fuel vendor analysis, it has been determined that the scram was not a result of THI. A review of APRM channel 2 signal output, active RCR loop driving flow, core flow, and core differential pressure at the time of the scram indicates the recirculation flow began to oscillate just prior to the scram. This oscillation appears to have caused the power oscillations that exceeded the GRA trip settings, since core flow and core differential pressure changes are synchronized with the RCR loop flow changes. No other parameters of interest appear to have contributed to the power oscillations at the time of the scram. It was determined that the power oscillations that exceeded the OPRM GRA trip setpoint were not caused by the high reactor noise level (i.e., reactor power oscillations) directly. The reactor noise level was higher as compared to Two Loop Operation (TLO) at full power and also while in SLO prior to raising power. The combination of operating in a condition where reactor noise was inherently high to begin with, coupled with the RCR loop flow varying momentarily in an increasing oscillatory manner caused the power oscillations to exceed the OPRM GRA trip setpoint. The analysis concluded that the OPRM trip was the result of externally induced core wide power oscillations caused by momentary oscillations in RCR loop 2B flow. Due to the high APRM noise band from SLO, the power oscillations matched the amplitude and period criteria for a GRA trip. The oscillations were not driven by THI, as the decay ratio was significantly less than the 0.8 threshold for THI to occur, there was no plant transient initiating event prior to the scram, and the power oscillations were not self-sustaining once the oscillations in RCR flow disappeared.

Two Operating Experience (OE) events are similar to the conditions which resulted in the BSEP SLO scram. The first was a Plant Hatch event, when reactor power was increased while in SLO, and increased fluctuations in jet pump flows, reactor level, and APRM noise bandwidth occurred. Plant Hatch conservatively reduced power to 44% power level, where it was held until the plant was shutdown for maintenance. It was determined that variations in operating parameters during SLO can be expected to be as high as twice those observed during normal TLO. SLO conditions introduce various thermal hydraulic effects that result in different level readings between instruments, however, it did not indicate any reduced margin to the trip setpoints. This OE was shared with BSEP Engineering, but it was determined that no action was needed since the concern was with the Period Based Detection Algorithm (PBDA) and the Plant Hatch amplitude setpoints were more conservative than BSEP and, therefore, more sensitive.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2006	-- 003 --	00	5 OF 6

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

EVENT DESCRIPTION (continued)

The second OE event was discussed in a Boiling Water Reactor Owners Group (BWROG) letter regarding increased potential for an OPRM trip in SLO, issued in April 2005, to the BWROG primary representatives. This discussed an event at Peach Bottom Atomic Power Station Unit 2 where an OPRM trip signal was received during SLO. The OPRM logic had not yet been activated so no plant actuations resulted, but the trip signal was unexpected. The event occurred with the reactor at 48% power and 49% core flow. Analysis determined that the trip was generated due to increased reactor noise that was two or more times that for TLO. It concluded this noise could be sensed by the OPRM system as oscillations indicative of instability. The purpose of the letter was to caution utilities of the increased potential for PBDA OPRM trip signal generation during SLO due to higher reactor noise levels relative to TLO. This OE was received from the BWROG and was not processed through the BSEP OE system. The System Engineer developed a briefing paper, which was distributed to the Control Room Operators and to the Reactor Engineers. This paper recommended interim actions to minimize time spent in SLO and maximize operator awareness of this issue. The Reactor Engineers incorporated the OE into their informal Reactor Engineer Shift Turnover which is updated each week. The OE information was not incorporated into Operations procedures or the lessons learned database. No specific action was identified by BSEP Engineering because the event involved a concern with PBDA settings and the Peach Bottom amplitude settings were more conservative than BSEP and, therefore, more sensitive.

EVENT CAUSE

The root cause of this event is inadequate incorporation of OE into plant procedures and training. Industry OE and BWROG OE did not result in the appropriate precautions and operating limitations for SLO being incorporated into plant procedures and training. BSEP review of the OE identified the need for action by Operations and Reactor Engineering, however the follow-up in April 2005 was handled informally, outside the formal OE process.

A contributing cause was that the analytical tools for determining the appropriate algorithm settings in the OPRM for Stability Option III are limited in their ability to predict their sensitivity to spurious scrams due to random noise, particularly in SLO. BSEP was concerned that the Option III strategy could increase the risk of spurious scrams, but it was concluded that this risk was acceptably small based upon the analysis of the risk, operating experience, and the improved method Option III provided for addressing THI concerns.

Another contributing cause was that the ODMs did not adequately evaluate the risk associated with extended SLO operation given that BSEP had only limited experience at low power in the OPRM enabled region. They focused on the risks related to a dual RCR seal failure on a pump, and did not fully consider all pertinent OE on SLO nor the increased risk of a spurious scram from extended operation in SLO.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2006	-- 003 --	00	6 OF 6

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

SAFETY ASSESSMENT

The safety significance of this condition is considered minimal. The RPS initiated the reactor scram as designed, vessel water level was promptly restored, and all appropriate safety system actuations functioned as expected. Operations personnel responded appropriately in accordance with procedures and ensured plant safety was not jeopardized. No actual THI condition existed and fuel thermal limits were not challenged.

CORRECTIVE ACTIONS

The following are corrective actions to prevent recurrence:

- Interim operating guidance which restricts operation in SLO was implemented.
- Additional procedurally specified operating margin and enhanced monitoring requirements to limit the risk of a scram when in SLO will be provided.
- The Core Operating Limits Report (COLR) Power/Flow maps will be revised to provide conservatism to the MELLLA line when operating in OPRM enabled region during SLO.
- A formal process will be developed to evaluate correspondence from sources outside of the Institute of Nuclear Power Operations (INPO) OE, such as BWROG organizations, Electric Power Research Institute (EPRI), and vendor information, to be captured in the BSEP procedures and processes.

The following are additional corrective actions:

- An Engineering Change to raise the Confirmation Density Algorithm (CDA) to prevent unnecessary alarms/trips from causing nuisance alarms on the Process Computer Event Alarm Log was implemented.
- OPRM trip settings will be evaluated to minimize spurious scrams and alarms.
- Operator training will be enhanced on THI, APRM noise, and the OPRM system upon evaluation of this event.

PREVIOUS SIMILAR EVENTS

A review of LERs and corrective action program condition reports which have occurred within the past three years has not identified any previous similar occurrences.

COMMITMENTS

No regulatory commitments are contained in this report.